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Abstract

FLUID DYNAMICS, STABILITY, AND VAPOR-LIQUID SLIP IN BOILING REACTOR SYSTEMS

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Recent progress in the study of two phase flow at government laboratories, universities and in industry is reviewed.

Studies at Argonne National Laboratory cover pressure drop and slip ratio in two phase flow through nozzles, slip ratios in horizontal and vertical flow, the effect of liquid viscosity on slip ratio, slip in two component systems, slip at low mass flow rates, vapor carryover problems, and a few selected instrumentation techniques for measuring void fractions.

Work reported from the General Electric Company deals with such problems as the prediction of voids in bubbling systems, boiling in horizontal flow, the steady state and transient response of void systems, the propagation of kinematic waves in two phase mixtures, the flow stability of boiling systems and steam separation studies.

With reference to the stability of boiling systems, a review is given of experiments on natural circulation flow stability, both in industry and in various universities. Two conclusions drawn from this review are (1) flow instabilities may arise from either the boiling section of the system, or the external parts of the system, or an interaction between the boiling section and the rest of the system; (2) no definite statements related to the onset of flow instability can be made unless the region of operation of the system is carefully specified, and the direction of change of the system variables is also specified.

Two principal areas studied by Allis Chalmers in regard to reactor core hydraulics are pressure drop and steam volume fraction (void fraction). Pressure drop studies conducted at 600 psi in support of the Pathfinder reactor

project showed that the parallel-rod geometry could be treated with the established two phase pressure drop correlations. An extensive study of void fractions in flow conditions similar to those found in "risers" is being conducted under the Euratom program. A limited study of void fractions in parallel-rod geometry was conducted for Pathfinder. In conjuction with the steam separation program, void fraction measurements were also taken on bubbling two phase mixtures. These data covered a range of pressures from 300 to 2000 psi and vessel diameters from 4 to 36 inches.

Studies of boiling water reactor stability at the Space Technology Laboratories have included laboratory measurements of the power-void transfer function in a simulated SPERT IA water coolant channel. These transfer function measurements with natural circulation showed that the divergent power oscillations observed in the SPERT IA reactor at 97C can be explained as a reactivity feedback instability, not as a purely hydrodynamic instability. As the power level in this channel is raised to approach the threshold of hydrodynamic instability, a resonance peak develops in the transfer function at 1 cycle/sec, the frequency at which a hydrodynamic oscillation eventually occurs. The resonance indicates that the void response to power modulation becomes dominated by the interaction between inlet flow and steam void, and when this feedback interaction becomes unstable a spontaneous hydrodynamic oscillation occurs.

An attempt was made to reproduce in the laboratory the destructive pressure pulses which occurred in the SPERT I and BORAX I reactor excursions. Transient electrical heating produced surface temperature rises of 700C in 150μsec with a stainless steel heated element in a geometry which modelled the unmolten SPERT I and BORAX I metal cores. The magnitudes of the pressure pulses produced by heat transfer to the water were about 800 psi, which is much less than was observed in the reactor excursions. Negative gauge pressure (vacuum) follows vapor blanketing in these thermal transients.